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OCT 24 2011

10CFR50.73

LR-N11-0307

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington DC 20555-001

SUPPLEMENTAL LER 311/2011-004-01
Salem Nuclear Generating Station Unit 2
Facility Operating License No. DPR-75
NRC Docket No. 50-311

SUBJECT: Automatic Reactor Trip Due to Trip of the 23 Reactor Coolant Pump

This Supplemental Licensee Event Report, "Automatic Reactor Trip Due to Trip of the 23 Reactor Coolant Pump," is being submitted pursuant to the requirements of the Code of Federal Regulations 10CFR50.73(a)(2)(iv)(A), "any event or condition that resulted in manual or automatic actuation of any of the systems listed in paragraph (a)(2)(iv)(B)."

The attached LER contains no commitments. Should you have any questions or comments regarding this submittal, please contact Mr. Brian Thomas at 856-339-2022.

Sincerely,

Carl J. Fricker
Site Vice President – Salem

Attachments (1)

IE22
NRR

OCT 24 2011

cc Mr. W. Dean, Administrator, Region I, NRC
Mr. R. Ennis, Licensing Project Manager – Salem, NRC
Mr. D. Schroeder, USNRC Senior Resident Inspector, Salem (X24)
Mr. P. Mulligan, Manager IV, NJBNE
L. Marabella, Corporate Commitment Tracking Coordinator
H. Berrick, Salem Commitment Tracking Coordinator

NRC FORM 366 (10-2010) U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2013

Estimated burden per response to comply with this mandatory collection request: 80 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Section (T-5 F53), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects.resource@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104). Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

LICENSEE EVENT REPORT (LER)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE
Automatic Reactor Trip Due to Trip of the 23 Reactor Coolant Pump

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
06	26	2011	2011	0 0 4	1	10	24	2011		DOCKET NUMBER

9. OPERATING MODE
1

10. POWER LEVEL
100%

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§: *(Check all that apply)*

- ☐ 20.2201(b)
- ☐ 20.2201(d)
- ☐ 20.2203(a)(1)
- ☐ 20.2203(a)(2)(i)
- ☐ 20.2203(a)(2)(ii)
- ☐ 20.2203(a)(2)(iii)
- ☐ 20.2203(a)(2)(iv)
- ☐ 20.2203(a)(2)(v)
- ☐ 20.2203(a)(2)(vi)
- ☐ 20.2203(a)(3)(i)
- ☐ 20.2203(a)(3)(ii)
- ☐ 20.2203(a)(4)
- ☐ 50.36(c)(1)(i)(A)
- ☐ 50.36(c)(1)(ii)(A)
- ☐ 50.36(c)(2)
- ☐ 50.46(a)(3)(ii)
- ☐ 50.73(a)(2)(i)(A)
- ☐ 50.73(a)(2)(i)(B)
- ☐ 50.73(a)(2)(i)(C)
- ☐ 50.73(a)(2)(ii)(A)
- ☐ 50.73(a)(2)(ii)(B)
- ☐ 50.73(a)(2)(iii)
- ☒ 50.73(a)(2)(iv)(A)
- ☐ 50.73(a)(2)(v)(A)
- ☐ 50.73(a)(2)(v)(B)
- ☐ 50.73(a)(2)(v)(C)
- ☐ 50.73(a)(2)(v)(D)
- ☐ 50.73(a)(2)(vii)
- ☐ 50.73(a)(2)(viii)(A)
- ☐ 50.73(a)(2)(viii)(B)
- ☐ 50.73(a)(2)(ix)(A)
- ☐ 50.73(a)(2)(x)
- ☐ 73.71(a)(4)
- ☐ 73.71(a)(5)
- ☐ OTHER

Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
Brian Thomas, Senior Compliance Engineer	(856) 339 -2022

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
B	EA	CBL5	A385	Y					

14. SUPPLEMENTAL REPORT EXPECTED

☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ☒ NO

15. EXPECTED SUBMISSION DATE

MONTH	DAY	YEAR

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

This report is being made in accordance with 10CFR50.73(a)(2)(iv)(A), "any event or condition that resulted in manual or automatic actuation of any of the systems listed paragraph (a)(2)(iv)(B)."

At approximately 1800 hours on June 26, 2011, the 23 Reactor Coolant Pump (RCP) tripped resulting in an automatic reactor trip on low flow in one reactor coolant loop above the P-8 permissive (36% power permissive). As expected, the 21, 22 and 23 Auxiliary Feedwater (AFW) pumps started on low steam generator level following the unit trip. Unit 2 was stabilized in Mode 3 at normal operating temperature and pressure with the 21, 22 and 24 RCPs in-service.

Salem Unit 2 tripped on low reactor coolant flow in the 23 reactor coolant loop. The low flow condition was the result of the trip of the 23 RCP due to two separate phase to ground faults. Both ground faults occurred at the termination to the motor where the cable shield interfaces with the stress tube of the termination kit. The insulation at this location was found to be brittle. Existing cable testing practices performed in accordance with industry standards would not detect this cable condition. Failure analysis determined that a partial discharge test or potentially destructive testing would have to be performed to detect the cable insulation degradation.

The 23 RCP motor was tested satisfactorily. The damaged sections of the RCP motor leads were removed and new terminations were made. Megger testing and visual examination of the 21, 22, and 24 RCP motor leads was performed. The post maintenance testing procedure will be revised. The RCP motor feeder cables for the Unit 1 and 2 RCPs will be replaced.

LICENSEE EVENT REPORT (LER)

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NARRATIVE

PLANT AND SYSTEM IDENTIFICATION

Westinghouse – Pressurized Water Reactor (PWR/4)

Reactor Coolant System / Reactor Coolant Pump {AB/P}

* Energy Industry Identification System {EIIS} codes and component function identifier codes appear as {SS/CCC}

IDENTIFICATION OF OCCURRENCE

Event Date: June 26, 2011

CONDITIONS PRIOR TO OCCURRENCE

Salem Unit 2 was in Mode 1. No additional structures, systems or components were inoperable at the time of the discovery that contributed to the event.

DESCRIPTION OF OCCURRENCE

At approximately 1800 hours on June 26, 2011, the 23 Reactor Coolant Pump (RCP) {AB/P} tripped resulting in an automatic reactor trip on low flow in one reactor coolant loop above the P-8 permissive (36% power permissive). As expected, the 21, 22 and 23 Auxiliary Feedwater (AFW) pumps started on low steam generator level following the unit trip. Unit 2 was stabilized in Mode 3 at normal operating temperature and pressure with the 21, 22 and 24 RCPs in-service.

CAUSE OF OCCURRENCE

Salem Unit 2 tripped on low reactor coolant flow in the 23 reactor coolant loop. The low flow condition was the result of the trip of the 23 RCP due to two separate phase to ground faults. Both ground faults occurred at the termination to the motor where the cable shield interfaces with the stress tube of the termination kit. The insulation at this location was found to be brittle. The reason for the severe hardening of the cable insulation under the stress relief tube was unable to be determined by laboratory testing. Existing cable testing practices performed in accordance with industry standards would not detect this cable condition. Failure analysis determined that a partial discharge test or potentially destructive testing would have to be performed to detect the cable insulation degradation.

PREVIOUS OCCURRENCES

A review of LERs at Salem Station dating back to 2008 did not identify any prior similar occurrences as a result of medium voltage cable degradation.

SAFETY CONSEQUENCES AND IMPLICATIONS

The loss of forced reactor coolant flow is a Condition II Event analyzed in Section 15.2.5 of the UFSAR. The UFSAR analysis states that the departure from nucleate boiling ratio (DNBR) will not decrease below the limiting value at any time during the transient and therefore no core safety limit is violated. The UFSAR analysis assumes a reactor trip on low RCS loop flow within 1.6 seconds of reduction in RCS loop flow with control rods beginning to drop in 2.6 seconds. Based on the review of post trip data the low flow reactor trip occurred in less than 1 second from the opening of the 23 RCP breaker and control rod drop occurred in less than 2 seconds. These response times were within the analyzed values therefore there were no safety consequences as a result of the reactor trip.

A review of this event determined that a Safety System Functional Failure (SSFF) as defined in NEI 99-02, Regulatory Assessment Performance Indicator Guidelines, did not occur. This event did not result in a condition that would have prevented the fulfillment of a safety function of a system needed to shutdown the reactor and maintain it in a safe shutdown condition, remove residual heat, control the release of radioactive material, or mitigate the consequences of an accident.

CORRECTIVE ACTIONS

1. The 23 RCP motor was tested satisfactorily. The damaged sections of the RCP motor leads were removed and new terminations were made.
2. Megger testing and visual examination of the 21, 22, and 24 RCP motor leads was performed with no immediate concerns identified.
3. Salem Unit 1 RCP motor feeder cables for the 11, 12, 13 and 14 RCP motors will be replaced during the 1R21 refueling outage.
4. Salem Unit 2 RCP motor feeder cables for the 21, 22, 23 and 24 RCP motors will be replaced during the 2R19 refueling outage

COMMITMENTS

No commitments are made in this LER.